Entergy

Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y. 10511-0249
Tel (914) 254-6700

John A. Ventosa
Site Vice President
Administration

NL-14-021

March 4, 2014

U.S. Nuclear Regulatory Commission
Document Control Desk
11545 Rockville Pike, TWFN-2 F1
Rockville, MD 20852-2738

SUBJECT: Licensee Event Report # 2014-001-00, "Automatic Reactor Trip as a Result of Steam Flow/Feedwater Flow Mismatch with Low 33 Steam Generator (SG) Water Level Due to the Failure of the 33 SG Feedwater Flow Controller"
Indian Point Unit No. 3
Docket No. 50-286
DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2014-001-00. The attached LER identifies an event where the reactor automatically tripped, which is reportable under 10 CFR 50.73(a)(2)(iv)(A). As a result of the reactor trip, the Auxiliary Feedwater System was actuated, which is also reportable under 10 CFR 50.73(a)(2)(iv)(A). This condition was recorded in the Entergy Corrective Action Program as Condition Report CR-IP3-2014-00054.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, Regulatory Assurance at (914) 254-6710.

Sincerely,

JAV/cbr

cc: Mr. William Dean, Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 3
Ms. Bridget Frymire, New York State Public Service Commission

IE22
NRR

NRC FORM 366 (01-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER)	

1. FACILITY NAME: INDIAN POINT 3	2. DOCKET NUMBER 05000-286	3. PAGE 1 OF 5

4. TITLE: Automatic Reactor Trip as a Result of Steam Flow/Feedwater Flow Missmatch with Low 33 Steam Generator (SG) Water Level Due to the Failure of the 33 SG Feedwater Flow Controller

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
1	06	2014	2014	001 -	00	3	04	2014	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
James Timone, Engineer, Engineering Systems Support	(914) 254-6733

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	JB	FC	N430	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

16. ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced type written lines)*

On January 6, 2014, the 33 Steam Generator (SG) steam flow (SF) Feedwater flow (FF) mismatch and SG level control deviation alarms annuniciated. Operators noticed the selected feedwater (FW) flow (FF) channel at zero and swapped to the alternate channel and noted both FW flow channels at zero. With the 33 SG level at approximately 10% and lowering, operators started actions to initiate a manual reactor trip (RT) when an automatic RT occurred on a SF/FF mismatch coincident with low 33 SG level. All control rods fully inserted and all required safety systems functioned properly with the exception of the intermediate range excore neutron flux detector N-35 which was undercompensated. The plant was stabilized in hot standby with decay heat being removed by the main condenser {SG}. The Auxiliary Feedwater System automatically started as expected due to SG low level from shrink effect. The Emergency Diesel Generators did not start as offsite power remained available and stable. Investigations determined the decreasing SG levels was due to reduced main FW flow as a result of a closed 33 Feedwater Regulating Valve (FRV). The closure of the 33 FRV-437 was due to no output from flow controller FC-437. The apparent cause was station personnel failed to apply lateral thinking and teamwork and question what other options were available besides procuring an identical controller. Corrective actions included replacing FC-437 and FC-417. A project to improve procurement and ease replacement of NUS modules will be developed. The event had no effect on public health and safety.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 3	05000-286	2014	- 001 -	00	2 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within the brackets {}.

DESCRIPTION OF EVENT

On January 6, 2014, while at 100% steady state reactor power, the 33 Steam Generator (SG) {AB} steam flow (SF) Feedwater Flow (FF) mismatch and SG level control deviation alarms annuniciated at approximately 21:14 hours, followed immediately by a 32 Main Boiler Feedwater Pump (MBFP) vibration alarm. Operators observed the MBFPs operation and realized they were responding to a reduction in feedwater (FW) flow. Operators noticed the selected feedwater (FW) flow (FF) channel at zero and swapped to the alternate channel and noted both FF channels at zero. With the 33 SG level at approximately 10% and lowering, operators started actions to initiate a manual reactor trip (RT) {JC} when an automatic RT occurred at approximately 21:15 hours, on a SF/FF mismatch coincident with low SG level. All control rods {AA} fully inserted and all required safety systems functioned properly with the exception of the intermediate range excore neutron flux detector N-35 which was undercompensated. The plant was stabilized in hot standby with decay heat being removed by the main condenser {SG}. The Auxiliary Feedwater System {BA} automatically started as expected due to SG low level from shrink effect. The Emergency Diesel Generators {EK} did not start as offsite power remained available and stable. There was no radiation release. Investigations determined the decreasing SG levels was due to reduced main FW flow as a result of a closed 33 Feedwater Regulating Valve (FRV) (FCV-437) {FCV}. The closure of the 33 FRV-437 was due to no output from flow controller FC-437 {FC}. The RT event was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP3-2014-00054. A post trip evaluation was initiated on January 7, 2014.

The investigation focused on response of the SG Water Level Control (SGWLC) system {JB} for maintaining level and MBFP speed. The SGWLC system consists of four three element control configurations, one for each SG, to control the position of its associated FRV. The SGWLC system senses steam flow and FW flow mismatch and deviation from level set point and sends a signal to the FRV positioners to modulate the FRVs.

Main FW regulating valve (FRV-33) BFD-FCV-437 is an air operated flow control globe valve (AOV) manufactured by Copes Vulcan {C635}, Model D-100-160 actuator and valve. The valve fails closed on a loss of air and has a Bailey Model AV-1 positioner (ABB Brown Boveri) {B455}. The valve's safety function is to close to terminate FW flow to the SG. The valve will close by venting air pressure on receipt of a SG High Level signal, a Safety Injection signal or a RT signal. The elements that encompass the control loop are the Controller, current to pneumatic (I/P) Converter, Positioner, and Actuator/valve. The controller provides a signal to the I/P based on steam flow, FW flow, and SG level offset. The I/P converts the controller signal to a pressure which modulates the positioner to move the valve to the demanded position.

The investigation into the cause of the 33 FRV going closed started by verifying that the 33 FRV stroked in manual control from the auto-manual station FIC-437. Subsequently, the 33 SG FW Flow controller {FC} FC-437 was verified as receiving the correct inputs. The investigation found that there was no output from FC-437, indicating that there was an internal failure of the controller. The 33 SG FW flow controller FC-437 {FC} is manufactured by NUS Corp {N430}.

Actual SG NR level is monitored and compared to the set point value and a level deviation output is produced be the level controller LC-437M and sent to the flow controller FC-437. The flow controller combines the level deviation input with the flow deviation it computes from steam flow and FW flow. The resulting auto signal is sent to the auto-manual station (FIC-437) which passes the signal to the 33 FRV when in auto mode. When in the manual mode, the output signal is determined by the operator at FIC-437.

A review of the history of FC-437 determined the replacement of FC-437 had been previously planned and scheduled for January 17, 2014, as part of the process instrumentation action plan and Maintenance Rule A1 action plan for the Unit 3 Reactor Protection System {JC} due to a known failure mechanism. The failure mechanism was first discovered upon failure of FC-447 on November 28, 2011 (CR-IP3-3011-05297). This failure mechanism applied to earlier vintage NUS modules that were installed in the 1990's and early 2000's that contain power supplies manufactured by Kepco {K078}. The Kepco power supplies contain aluminum electrolytic capacitors that have an expected combined shelf life and service life of 15 years. Both FC-437 and FC-417 were beyond this expected life. The newer vintage NUS modules manufactured after 2007 contain Ensign {E280} power supplies that utilize tantalum capacitors with an expected life of 40 years.

Work Orders (WO) to replace FC-437 and FC-417 were initiated on December 19, 2011 as part of an extent of condition for the FC-447 failure. These WO's were being tracked under CR-IP3-2011-05297 Corrective Action (CA) 9. CA-9 was initially due on May 24, 2012 but was extended when the WO's were re-scheduled due to lack of material (replacement of three element controllers). The CA was closed based on the replacement controllers were due from NUS on December 26, 2012 and the WO's were being tracked as Unit Reliability Team commitments. The replacement WO's had been re-scheduled multiple times (5) due to material issues. The material issues regarding procurement of replacement controller had several contributing factors. There is a significant lead time to procure new controllers from NUS. The initial order of replacement controllers were not received until May 2013 which was past the original scheduled date and when received it was discovered generic controllers were received rather than the special three element controllers that were required for the application. Engineering reviewed this configuration issue and determined the NUS parts Catalog ID had been previously changed to a generic configurable NUS controller instead of a special three element controller specifically configured for the FC-417 through FC-447 application. On October 2013, a meeting was convened to determine the best course of action. It was concluded the best option was to send the controllers removed from FC-427 and FC-447 to NUS for refurbishment. The refurbishment by NUS was scheduled to be complete by January 17, 2014.

A plan was being worked in parallel for replacement of identical Unit 2 controllers (FC-417 through FC-447) during the upcoming Unit 2 refueling outage in February 2014. To procure replacement controllers for Unit 2, four generic controllers had been removed from stock and shipped to NUS to be converted to special three element controllers and were received at the site on January 6, 2014. After the RT it was decided to replace both the failed FC-437 and FC-417. During the process of calibrating the controllers, Instrumentation & Control (I&C) personnel noticed that the converted controllers were still generic controllers and had not been converted to special three element controllers by NUS. NUS had only configured them for the three element controller application. An engineering evaluation concluded the configured controllers (NUS PIDA700-5E) which had a set point dial and deviation meter were acceptable to use with internal jumpers installed to bypass the set point dial so that it is not capable of affecting the controller function.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

An extent of condition investigation determined the FC-437 controller was a known single point of vulnerability that was past the expected 15 year life for its internal power supply. The other NUS controllers that are a single point of vulnerability at Unit 3 have all been replaced since 2009 with new NUS controllers that contain a new style power supply that has a 40 year life. The remaining NUS controllers that are single point of vulnerability are at Unit 2 and are scheduled for replacement in the 2014 refueling outage. The Unit 2 controllers do not have an elevated risk because they are still within their 15 year life. There remain other NUS modules at both Unit 2 and 3 that are beyond their 15 year life and are scheduled for replacement but are in low critical applications. Work Orders were prepared to replace the remaining SG FCs, LC's and auto-manual stations at unit 2 in the 2014 refueling outage.

The Cause of Event

The direct cause of the RT was lowering SG levels and the inability to maintain SG levels. The decreasing 33 SG level was due to reduced FW flow from the closure of the 33 FRV-437 as a result of the failure of flow controller FC-437.

The apparent cause was station personnel exhibited tunnel vision to address the FW controller aging power supply vulnerability. Personnel focused on procuring an identical special three element controller as a replacement. The station knew there was a vulnerability and that the controller was past its expected 15 year life and was a single point vulnerability. However, addressing its replacement was allowed to be re-scheduled five times without applying lateral thinking to solve what was believed to be a parts problem as no special controllers were available. Personnel failed to apply lateral thinking and teamwork and question what other options were available besides procuring an identical controller. Subsequent to the RT, a solution of replacing the special controller with a generic controller and to provide a bypass of the set point dial via installation of internal jumpers was identified.

Corrective Actions

The following corrective actions have been or will be performed under the Corrective Action Program (CAP) to address the causes of this event.

- SG 33 flow controller FC-437 was replaced as was FC-417.
- A project to improve procurement and ease replacement of NUS modules will be developed.

Event Analysis

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply for this event include the Reactor Protection System (RPS) including RT and AFWS actuation.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

This event meets the reporting criteria because an automatic RT was initiated at 21:15 hours, on January 6, 2014, and the AFWS actuated as a result of the RT. On January 6, 2014, a 4-hour non-emergency notification was made to the NRC at 21:55 hours, for an actuation of the reactor protection system {JC} while critical and included an 8-hour notification under 10CFR50.72(b)(3)(iv)(A) for a valid actuation of the AFW System (Event Log #49698). As all primary safety systems functioned properly there was no safety system functional failure reportable under 10CFR50.73(a)(2)(v).

Past Similar Events

A review was performed of the past three years for Licensee Event Reports (LERs) reporting a RT as a result of main FW reduction. No LERs were identified that reported a RT due to a FW events.

Safety Significance

This event had no effect on the health and safety of the public.
There were no actual safety consequences for the event because the event was an uncomplicated reactor trip with no other transients or accidents. Required primary safety systems performed as designed when the RT was initiated. The AFWS actuation was an expected reaction as a result of low SG water level due to SG void fraction (shrink), which occurs after a RT and main steam back pressure as a result of the rapid reduction of steam flow due to turbine control valve closure.

There were no significant potential safety consequences of this event. The Reactor Protection System (RPS) is designed to actuate a RT for any anticipated combination of plant conditions to include low SG level. The reduction in SG level and RT is a condition for which the plant is analyzed. A low water level in the SGs initiates actuation of the AFWS. Redundant safety SG level instrumentation was available for a low SG level actuation which automatically initiates a RT and AFWS start providing an alternate source of FW. The AFW System has adequate redundancy to provide the minimum required flow assuming a single failure. The analysis of a loss of normal FW (UFSAR Section 14.1.9) shows that following a loss of normal FW, the AFWS is capable of removing the stored and residual heat plus reactor coolant pump waste heat thereby preventing either over pressurization of the RCS or loss of water from the reactor. In addition, Operators for this event anticipated a possible low SG level and could have initiated a manual RT. The manual actuating devices are independent of the automatic trip circuitry and are not subject to failures which make the automatic circuitry inoperable. There are two manual trip buttons, one located on flight panel FCF and the other on safeguards supervisory panel SBF2. Either one of these buttons will directly energize the trip coils of the reactor trip and bypass breakers in addition to de-energizing the undervoltage coils of the reactor trip and bypass breakers. For this event, rod control was in automatic and all rods inserted upon initiation of a RT. The AFWS actuated and provided required FW flow to the SGs. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.